DEPARTMENT OF LABOR

Employment and Training Administration

[TA–W–65,339]

Pentagon Technologies Group, Inc. Portland, OR; Notice of Termination of Investigation

Pursuant to Section 221 of the Trade Act of 1974, as amended, an investigation was initiated on February 23, 2009 in response to a worker petition filed by a company official on behalf of workers of Pentagon Technologies Group, Inc., Portland, Oregon.

The petitioner has requested that the petition be withdrawn. Consequently, the investigation has been terminated.

Signed at Washington, DC, this 24th day of February 2009.

Richard Church,

Certifying Officer, Division of Trade Adjustment Assistance.

[FR Doc. E9–5050 Filed 3–9–09; 8:45 am]

BILLING CODE 4510–FN–P

DEPARTMENT OF LABOR

Employment and Training Administration

[TA–W–65,299]

United States Steel Great Lakes Works, Ecorse, MI; Notice of Termination of Investigation

Pursuant to Section 221 of the Trade Act of 1974, as amended, an investigation was initiated on February 19, 2009 in response to a petition filed by the United Steelworkers of America, Local 1299 on behalf of workers of United States Steel Great Lakes Works, Ecorse, Michigan.

The petitioning group of workers is covered by an earlier petition (TA–W–64,773) filed on December 19, 2008 that is the subject of an ongoing investigation for which a determination has not yet been issued. Further investigation in this case would duplicate efforts and serve no purpose; therefore the investigation under this petition has been terminated.

Signed in Washington, DC, this 24th day of February 2009.

Linda G. Poole,

Certifying Officer, Division of Trade Adjustment Assistance.

[FR Doc. E9–5048 Filed 3–9–09; 8:45 am]

BILLING CODE 4510–FN–P

DEPARTMENT OF LABOR

Office of the Assistant Secretary for Veterans' Employment and Training

The Advisory Committee on Veterans' Employment, Training and Employer Outreach (ACVETEO); Notice of Open Meeting

The Advisory Committee on Veterans' Employment, Training and Employer Outreach (ACVETEO) was established pursuant to Title II of the Veterans' Housing Opportunity and Benefits Improvement Act of 2006 (Pub. L. 109–233) and Section 9 of the Federal Advisory Committee Act (FACA) (Pub. L. 92–462, Title 5 U.S.C. app.II). The authority of the ACVETEO is codified in Title 38 U.S. Code, Section 4110.

The ACVETEO is responsible for assessing employment and training needs of veterans; determining the extent to which the programs and activities of the U.S. Department of Labor meet these needs; and assisting to conduct outreach to employers seeking to hire veterans. The ACVETEO will conduct a business meeting on Friday, March 20, 2009 from 8:30 a.m. to 3:30 p.m., at the Omni Hotel, 401 Chestnut Street, second floor meeting room, Philadelphia, PA. The ACVETEO will discuss programs to assist veterans seeking employment and to raise employer awareness as to the advantages of hiring veterans, with special emphasis on employer outreach and wounded and injured veterans.

Individuals needing special accommodations should notify Margaret Hill Watts at (202) 693–4744 by March 9, 2009.

Signed in Washington, DC, this 2nd day of March 2009.

John M. McWilliam,

Deputy Assistant Secretary, Veterans' Employment and Training Service.

[FR Doc. E9–4915 Filed 3–9–09; 8:45 am]

BILLING CODE 4510–79–P

NUCLEAR REGULATORY COMMISSION

[NRC–2009–0100]

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 12, 2009, to February 25, 2009. The last biweekly notice was published on February 24, 2009 (74 FR 8281).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rulemaking, Directives and Editing Branch, TWB–05–B01M, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner/requestor to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated on August 28, 2007 (72 FR 49139). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *hearingdocket@nrc.gov*, or by calling (301) 415–1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at *http://www.nrc.gov/site-help/e-submittals/install-viewer.html*. Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html*.

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html*. A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The EIE system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the ''Contact Us'' link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html* or by calling the NRC electronic filing Help Desk, which is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays. The electronic filing Help Desk can be contacted by telephone at 1–866–672–7640 or by e-mail at *MSHD.Resource@nrc.gov*.

Participants who believe that they have a good cause for not submitting documents electronically must file a motion, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, *Attention:* Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, *Attention:* Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the Presiding Officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp*, unless excluded pursuant to an order of the Commission, the Atomic Safety and Licensing Board, or a Presiding Officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

For further details with respect to this amendment action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendment request: January 15, 2009.

Description of amendment request: The amendments would modify Technical Specifications (TSs) 3.3.10, 3.6.7, and 5.6.6 to delete the requirements related to hydrogen recombiners and hydrogen monitors. The proposed TS changes would support implementation of the revisions to 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' that became effective on October 16, 2003. The proposed changes are consistent with Revision 1 of the NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.''

The NRC staff issued a notice of opportunity for public comments on TSTF–447, Revision 1, published in the **Federal Register** on August 2, 2002 (67 FR 50374), soliciting comments on a model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination for the elimination of requirements for hydrogen recombiners, and hydrogen and oxygen monitors from TS. Based on its evaluation of the public comments received, the NRC staff made appropriate changes to the models and included final versions in a notice of availability published in the **Federal Register** on September 25, 2003 (68 FR 55416), regarding the adoption of TSTF–447, Revision 1, as part of the NRC's consolidated line item improvement process (CLIIP).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC adopted by the licensee is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG [Regulatory Guide] 1.97 Category 1 is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents.

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3 and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs [severe accident management guidelines], the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from Any Previously Evaluated

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the [Three Mile Island], Unit 2 accident, can be adequately met without reliance on safety-related hydrogen monitors.

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

The NRC staff has reviewed the analysis adopted by the licensee analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves NSHC.

Attorney for licensee: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072–2034.

NRC Branch Chief: Michael T. Markley.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: September 29, 2008.

Description of amendment request: The proposed changes would revise the TMI–1 technical specifications (TSs) to reflect design changes resulting from the planned control rod drive control system (CRDCS) digital upgrade project. In addition, the proposed amendment would revise the TS to remove all references to the axial power shaping rods (APSRs) to reflect changes resulting from their proposed elimination from the TMI–1 reactor.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below, with U.S. Nuclear Regulatory Commission (NRC) staff edits in brackets:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed license amendment modifies the Technical Specifications (TSs) to incorporate new TS requirements associated with the new Digital Control Rod Drive Control System (DCRDCS) and an evaluation to permanently remove the Axial Power Shaping Rods (APSRs) from the reactor core.

The proposed license amendment will continue to ensure reliability and operability of the control rod drive Reactor Trip Breakers (RTBs) to perform their safety function of tripping the reactor. The existing channel independence, separation and performance requirements of the RTBs and the Reactor Protection System (RPS) response time are retained for the new configuration. The RTB design was reviewed for credible common mode failures and no credible common mode failures were identified that would prevent the breakers from performing the reactor trip function. Reliable RTBs and their associated support circuitry provide assurance that a reactor trip will occur when initiated. The planned DCRDCS modification upgrades the relay-based Control Rod Drive Control System (CRDCS) to a solid state programmable DCRDCS using single rod power supplies assigned to each of the 61 Control Rod Drives (CRDs). The new components will meet the same design requirements (i.e., seismic, environmental, quality, separation, single failure criteria) as the existing components in the CRDCS/RPS interface. The DCRDCS modification will improve the reliability of the system by resolving age-related degradation issues and replacing obsolete equipment.

Malfunction of the CRD control system (or operator error) is an initiator of the startup and rod withdrawal accidents. The new DCRDCS meets the design requirements of the original system including redundancy of critical functions, isolation from safety related systems, reactivity rate limit, and single failure requirements. Electrical ratings, heat loading, structural and environmental aspects have been verified to be acceptable. Therefore, there is no increase in the frequency of occurrence or probability of a malfunction of equipment important to safety. The DCRDCS is not required for accident mitigation, post accident response or offsite release mitigation. The action of the RPS to trip the RTBs, to remove power from the control rods, and drop the rods into the core, remains independent of the DCRDCS. Therefore, there is no increase in the consequences or probability of occurrence of an accident previously evaluated.

The modified Diverse Scram System (DSS) design utilizes the same power sources as the existing DSS, which are independent of reactor trip (i.e., RPS) related power sources. There is no change to the DSS logic circuitry. The DSS sensors and trip setpoint remains unchanged. Updated Final Safety Analysis (UFSAR) Section 7.1.5.4 indicates that: ''The DSS provides an independent method of automatically tripping the reactor in the event the RPS related reactor trip system fails. It is designed in accordance with the Anticipated Transient Without Scram (ATWS) rule and, as such, its critical features are independence and diversity from the reactor trip system and emphasis on not failing in a tripped state.'' However, DSS is not safety related and is not credited in any safety analysis in UFSAR Chapter 14, ''Safety Analysis.'' The assumed DSS response time increase from 1.0 second to 2.0 seconds has been evaluated and the results of the analysis concluded that the original acceptance criteria are maintained. Therefore, the proposed change to the DSS [is not adverse and] does not increase the consequence of an ATWS event.

The proposed license amendment will continue to ensure the reliability and operation of the reactor core. Analyses have shown that the core designs employed at TMI–1 are stable with respect to axial oscillations and that xenon oscillations initiated during power transients are naturally damped or can be manually suppressed using regulating control rods (i.e., Control Rod Group 7 (CRG–7)). Actual operating experience at TMI–1 bears out the analysis conclusions that adequate axial imbalance control can be maintained using coordinated movements of CRG–7 [and] timed water additions. A review of the TMI–1 safety analyses found no mention or credit for APSRs in any of the events analyzed for TMI–1, and safety analysis assumptions are verified to bound key core parameters for each reload with explicit accounting for the presence of (or lack of) APSRs in the core. Therefore, there is no affect of APSRs on transient analyses, as APSR positions do not change in the event of a reactor trip.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. Rather, the CRDCS/RPS interface (i.e., RTBs) is used to mitigate the consequences of an accident that has already occurred. The proposed TS changes do not affect the mitigating function of this system. The failure of any one RTB will not inhibit the reactor trip function. The modification interfaces with the DSS, which mitigates the ATWS event, but the interface function remains the same.

A Failure Modes and Effects Analysis (FMEA) was performed on the DCRDCS design to determine if adverse effects (i.e., loss of reactor control, uncontrolled rod withdrawal, reactor trip, or prevention of reactor trip) could result from the credible failure of a single component. The FMEA concluded that no credible single component failure would cause a total loss of reactor control, an uncontrolled rod withdrawal, a reactor trip, or prevent a reactor trip. All operation critical to the safe and effective performance of the DCRDCS maintained sufficient redundancy such that no credible single failure could compromise the design functionality.

The APSRs' original function was to control any reactor core tendency towards axial oscillations resulting from xenon instabilities that could occur for certain early reactor core designs (*i.e.*, rodded core designs). More recent non-rodded feed-and-bleed core designs have been shown to be self-dampened with respect to axial xenon oscillations such that APSRs have not been moved at TMI–1 for axial power control since 1994, and have been withdrawn from the reactor core since Fall 2005 with Core Operating Limits Report limits preventing insertion, consistent with AREVA reload methods.

Use of [CRG–7] has been shown to adequately suppress axial xenon oscillations.

The proposed changes to the CRDCS and APSRs and associated TS changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed TS changes do not adversely impact any plant safety limits, setpoints, response times, or design parameters. The changes do not negatively affect the fuel, fuel cladding, reactor coolant system, or containment integrity [under normal, transient or accident conditions].

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. Bradley Fewell, Esquire, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, (TMI–1) Dauphin County, Pennsylvania

Date of amendment request: November 6, 2008.

Description of amendment request: The proposed amendment would modify the TMI–1 Technical Specifications (TS), to replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on dose equivalent Xenon-133 (DEX) and would take into account only the noble gas activity in the primary coolant. The completion time for DEX being out of specification would be increased to match the action time requirements for the dose equivalent Iodine-131 (DEI) specification. In addition, the current DEI definition would be revised to allow the use of additional options for determining thyroid dose conversion factors. This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–490. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 20, 2006 (71 FR 67170), on possible amendments concerning TSTF–490, including a model safety evaluation and model no significant hazards (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 15, 2007 (72 FR 12217). The licensee affirmed the applicability of the following NSHC determination in its application dated November 6, 2008.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

Criterion 1: The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

Reactor coolant specific activity is not an initiator for any accident previously evaluated. The Completion Time when primary coolant gross activity is not within limit is not an initiator for any accident previously evaluated. The current variable limit on primary coolant iodine concentration is not an initiator to any accident previously evaluated. As a result, the proposed change does not significantly increase the probability of an accident. The proposed change will limit primary coolant noble gases to concentrations consistent with the accident analyses. The proposed change to the Completion Time has no impact on the consequences of any design basis accident since the consequences of an accident during the extended Completion Time are the same as the consequences of an accident during the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

Criterion 2: The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Accident Previously Evaluated

The proposed change in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter. The change does not create the potential for a new or different kind of accident from any previously calculated.

Criterion 3: The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change revises the limits on noble gas radioactivity in the primary coolant. The proposed change is consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses. Based upon the reasoning presented above, the requested change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the analysis and based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. Bradley Fewell, Esquire, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: October 9, 2008.

Description of amendment request: The proposed changes would revise the existing Three Mile Island (TMI), Unit 1, technical specifications (TSs) relating to the steam generator (SG) tube surveillance program. The proposed changes reflect the planned installation of replacement SGs and specifically address the new thermally treated Alloy 690 tubing design of the replacement SGs. Removal of sections of the TSs that are not applicable to the replacement SGs are proposed.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below, with U.S. Nuclear Regulatory Commission (NRC) staff edits in brackets:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the Technical Specifications (TSs) for the TMI, Unit 1 Steam Generator (SG) Program recognize that the TMI, Unit 1 SGs are being replaced and the standard industry performance criteria documented in [Technical Specification Task Force (TSTF) Traveler,] TSTF–449[,] for Alloy 690-tubed SGs will apply. These changes eliminate criteria that were established to reflect the condition and materials of the current TMI, Unit 1 SGs, and add the requirements for inspection of Alloy 690-tubed SGs from TSTF–449.

With these proposed TS changes, the operational primary-to-secondary leakage rate limit established for the original TMI, Unit 1 SGs is replaced with the standard industry primary-to-secondary leakage rate limit. The standard industry limit is that limit provided in TSTF–449. The current, reduced limit in the TMI, Unit 1 TS was implemented in response to upper tubesheet tube expansion degradation, and repairs, in the original TMI, Unit 1 SGs. A reduced limit is not required for the replacement SGs since they are fabricated from advanced materials and [will not be] subjected to the degradation mechanisms that influenced the original TMI, Unit 1 SGs. Thus, reverting to the standard industry limit is appropriate. The slightly higher, industry standard, leak rate limit is still low enough to provide assurance that the probability of tube ruptures, or of rapidly propagating tube leaks, remains acceptably low. Thus, the probability of a previously evaluated accident is not increased.

The installation of the new SGs, with improved materials, will decrease the consequences of SG related accidents. The removal of accident-induced leakage attributable to the current degradation mechanisms from TS 6.19.c.1.b [provides a reduction in the] accident induced leakage limit to 1 gpm per SG. SG accident-induced leakage is proportional to dose; a lower accident-induced leakage limit will result in a lower dose than previously evaluated accident consequences.

The proposed change to replace the 90-day report with a report required within 180 days is a change to an administrative requirement and does not affect the probability or consequences of an accident. The 180-day period is now industry ''standard'' practice per TSTF–449.

These changes continue to provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). With the proposed changes, the SG performance criteria (based on tube structural integrity, accident-induced leakage, and operational leakage) and SG Program are updated to reflect the replacement SGs while remaining consistent with TSTF–449.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident that was previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed TS changes recognize an improvement in SG design as a result of SG replacement. The replacement SGs contain a number of design improvements with respect to the plant's original SGs. However, even with the design improvements, the replacement SGs are very similar to the original SGs and new types of accidents are not created. There are no new design functions for the Alloy 690 tubing in the replacement SGs. The proposed new leakage and inspection requirements are the standard industry requirements for Alloy 690 tubing.

Primary-to-secondary leakage monitoring equipment is not affected by the proposed changes, and primary-to-secondary leakage will continue to be monitored to ensure it remains within current accident analysis assumptions and limits. The proposed changes implement the industry ''standard'' TSTF–449 primary-to-secondary leak limits for the plant's Alloy 690-tubed replacement SGs. No new types of primary-to-secondary leak accidents are created.

The proposed change to replace the 90-day report with a report required within 180 days is a change to an administrative requirement and does not create a new or different kind of accident. The 180-day period is now industry ''standard'' practice per TSTF–449.

Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The SG tubes in pressurized water reactors [PWRS] are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. The SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

SG tube integrity is a function of the design, environment, and physical condition of the tubing. The proposed changes do not affect the operating environment but do recognize the improved tube material as a result of replacing the SGs. The proposed TS changes for inspection, repair, and leakage requirements are consistent with industry codes and standards for replacement SGs with Alloy 690 tubing material. The requirements established by the SG Program are consistent with those in the applicable design codes and standards. The proposed changes update the requirements in the current TSs to reflect SG replacement.

The proposed TS changes include a change to the current TS limit on primary-to-secondary leakage of 144 GPD [gallons per day] that was established in the 1980s due to SG tube degradation. The basis for this limit will no longer be applicable with the installation of replacement SGs. The proposed limit of 150 gallons per day of primary-to-secondary leakage through any one SG is ''standard'' for the U.S. PWR industry. This limit is based on operating experience with SG tube degradation mechanisms that result in leakage and provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. Further, if it is not practical to assign the leakage to an individual SG, all the primary-to-secondary leakage is conservatively assumed to be from one SG. This operational leakage rate criterion, in conjunction with the implementation of the SG Program, is an effective measure for minimizing the frequency of SG tube ruptures. [Additionally, this TS requirement is significantly less than the conditions assumed in the safety analysis.]

The proposed change to replace the 90-day report with a report required within 180 days is a change to an administrative requirement and does not affect the margin of safety. The 180-day period is now industry ''standard'' practice per TSTF–449.

For the above reasons, the margin of safety is not reduced.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. Bradley Fewell, Esquire, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: December 4, 2008.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.9.3, ''Containment Penetrations,'' to permit refueling operations with both personnel airlock doors open under administrative control. Nuclear Regulatory Commission (NRC) review and approval of a revised non loss-of-coolant accident (LOCA) gas gap fractions and fuel-handling accident (FHA) using the revised gap fractions and a shorter decay time of 72 hours will be necessary to support this license amendment.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

There are three separate items requiring NRC approval in the licensee's application. The licensee has submitted a plant-specific analysis to revise the non-LOCA gas gap fractions. Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' includes Table 3, ''Non-LOCA Fraction of Fission Product Inventory in Gap.'' The Ginna licensee has determined that a small number of fuel rods may exceed the peak power and burnup criteria of Table 3 thus necessitating the plant-specific analysis. The new non-LOCA gap fractions are considered a methodology change thus requiring NRC review and approval.

The Ginna FHA currently assumes that fuel movement will not occur prior to 100 hours following reactor shutdown. The licensee has submitted a revised FHA that assumes both the new gas gap fractions discussed above and only 72 hours of decay time prior to fuel movement. The revised FHA must also be reviewed and approved by NRC.

The proposed change to TS 3.9.3, which would permit refueling operations with both personnel airlock doors open under administrative control, impacts the release pathway for the FHA. The proposed TS change requires NRC review and approval.

The proposed changes to the gas gap fractions and the FHA represent analytical changes and do not increase the probability of an accident previously evaluated. The change to TS 3.9.3 introduces a new release pathway for the FHA and does not increase the probability of an FHA or any other accident previously evaluated.

The change in analyzed decay time and the non-LOCA gap fractions result in an increase in the estimated dose to the control room and off-site receptors and, upon approval, will become the analyses of record. However, the increase in dose is within regulatory limits so that the changes do not represent a significant increase in the consequences of the FHA or any other accident previously evaluated. The proposed change to TS 3.9.3 introduces a new release pathway for the FHA. However, control room and offsite dose calculations are bounded by the release pathway from the equipment hatch. As a result, the proposed change to TS 3.9.3 does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the probability or consequences of an accident previously evaluated will not be significantly increased.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes in analyzed decay time and the non-LOCA gap fractions only impact design inputs to the FHA. The proposed change to TS 3.9.3 only impacts isolation requirements during refueling operations within the containment. The only accident which could result in a significant release of radioactivity in the plant mode where refueling is possible is the FHA. No other initiators or accident precursors are created by this change.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident not previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The change in analyzed decay time and the non-LOCA gap fractions result in an increase in estimated dose to the control room and off site receptors. However, the dose remains within regulatory guidelines and limits with adequate margin. The proposed change to TS 3.9.3 introduces a new release pathway for the FHA which is bounded by the release pathway through the equipment hatch.

Therefore, the proposed change does not involve a significant reduction in the margin of safety. Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Carey Fleming, Sr. Counsel—Nuclear Generation, Constellation Group, LLC, 750 East Pratt Street, 17th Floor, Baltimore, MD 21202.

NRC Branch Chief: Mark G. Kowal.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.22(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by email to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: November 13, 2007, as supplemented by letter dated February 18, 2009.

Description of amendment request: The amendment deletes Technical Specification (TS) Section 6.5 and its associated subsections relating to the Review and Audit function, as well as correcting several administrative items. Additionally, the amendment implements changes to correct minor errors in TS Tables 3.1.1, 4.1.1, and 4.1.2.

Date of issuance: February 24, 2009.

Effective date: As of its date of issuance, and shall be implemented within 60 days.

Amendment No.: 273.

Facility Operating License No. DPR–16: The amendment revised the License and Technical Specifications.

Date of initial notice in ***Federal Register****:* April 8, 2008 (73 FR 19108). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 2009.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of application for amendment: November 13, 2007, supplemented by letters dated September 29, 2008, and February 18, 2009.

Brief description of amendment: The amendment deletes Technical Specification (TS) Section 6.5 and its associated subsections relating to the Review and Audit function, as well as correcting several administrative items. The administrative items involve: correcting typographical errors, providing improved TS figure legibility, updating the description of the installed spent fuel pool storage locations, removing references to deleted TS sections, and correcting an error in the labeling of outfalls on the TMI site drawing.

Date of issuance: February 24, 2009.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 269.

Facility Operating License No. DPR–50. Amendment revised the license and the technical specifications.

Date of initial notice in ***Federal Register****:* April 8, 2008 (73 FR 19109). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 2009.

No significant hazards consideration comments received: No.

Duke Energy Carolinas, LLC, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of application for amendments: January 20, 2009.

Brief description of amendments: The amendment revised Technical Specification Surveillance Requirement (SR) 3.3.1.4 frequency. SR 3.3.1.4 is a Trip Actuating Device Operational Test of the reactor trip breakers and reactor trip bypass breakers.

Date of issuance: February 13, 2009.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment No.: 242.

Facility Operating License No. NPF–52: The amendment revised the license and the technical specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination by February 28, 2009. No comments have been received to date. However, the notice also provided an opportunity to request a hearing by March 30, 2009, but indicated that if the Commission make a final NSHC determination, any such hearing would take place after issuance of the amendment.

Date of initial notice in ***Federal Register****:* January 28, 2009 (74 FR 4986). The supplement dated February 5, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 2009.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: November 25, 2008.

Brief description of amendment: The amendment would revise Appendix A of Technical Specifications (TSs), as they apply to the spent fuel pool storage requirements in TS Section 3.7.16 and the criticality requirements for the Region I spent fuel pool and north tilt pit fuel storage racks, in TS Section 4.3.1.1.

Date of issuance: February 6, 2009.

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment No.: 236.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* January 2, 2009 (74 FR 123).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 2009.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: July 21, 2008.

Brief description of amendment: The amendment supports a proposed change to the in-service inspection program that is based on topical report WCAP–16168–NP–A, Revision 2, ''Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval.'' In the referenced safety evaluation of the topical report, the NRC required licensees to amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a) be submitted for NRC staff review and approval within one year of completing the required reactor vessel weld inspection. Entergy Nuclear Operations, Inc., added a new license condition to provide this information.

Date of issuance: February 11, 2009.

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment No.: 237.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* November 4, 2008 (73 FR 65690). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 11, 2009.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: March 1, 2007, as supplemented by letters dated September 5 and September 21, 2007, February 14, 2008, and January 19 and February 20, 2009.

Brief description of amendment: The changes revised the allowable values in the Grand Gulf Nuclear Station, Unit 1, Technical Specification Tables 3.3.5.1–1 and 3.3.5.2–1 for the Condensate Storage Tank (CST) low level setpoints for the High Pressure Core Spray and Reactor Core Isolation Cooling suction swap from the CST to the Suppression Pool.

Date of issuance: February 25, 2009.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No: 181.

Facility Operating License No. NPF–29: The amendment revised the Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register**:* May 8, 2007 (72 FR 26176). The supplements dated September 5 and September 21, 2007, February 14, 2008, and January 19 and February 20, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 2009.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3 (Waterford 3), St. Charles Parish, Louisiana

Date of amendment request: August 16, 2007, as supplemented by letter dated January 8, 2009.

Brief description of amendment: The amendment added a new license condition on the control room envelope (CRE) habitability program; revised the TS requirements related to the CRE habitability in TS 3.7.6, ''Control Room Emergency Air Filtration System—Operating,'' TS 3.7.6.2, ''Control Room Emergency Air Filtration System—Shutdown,'' and TS 3.7.6.5, ''Control Room Isolation and Pressurization''; and established a CRE habitability program in TS Section 6.5, ''Administrative Controls—Programs.'' These changes are consistent with the NRC-approved Industry/TS Task Force (TSTF) Traveler TSTF–448, Revision 3, ''Control Room Habitability.'' The availability of this TS improvement was published in the **Federal Register** on January 17, 2007 (72 FR 2022), as part of the Consolidated Line Item Improvement Process.

Date of issuance: February 20, 2009.

Effective date: As of the date of issuance and shall be implemented 120 days from the date of issuance.

Amendment No.: 218.

Facility Operating License No. NPF–38: The amendment revised the Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register**:* September 25, 2007 (72 FR 54473).

The supplemental letter dated January 8, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 2009.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois.

Exelon Generation Company, LLC, Docket No. 50–461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 1, 2 and 3, Grundy County, Illinois.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois.

Exelon Generation Company, LLC, Docket No. 50–352 and No. 50–353, Limerick Generating Station, Unit 1 and 2, Montgomery County, Pennsylvania.

Exelon Generation Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station,Units 2 and 3,York and Lancaster Counties, Pennsylvania.

Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania.

Date of application for amendments: February 28, 2008.

Brief description of amendments: The amendment incorporates Technical Specification Task Force Change

Traveler No. 308, Rev. 1, ''Determination of Cumulative and Projected Dose Contributions in the Radioactive Effluent Controls Program (RECP),'' which clarified the existing wording in the RECP technical specification to reflect the intent of Generic Letter 89–01, ''Implementation of Programmatic and Procedural Controls for radiological Effluent Technical Specifications (RETS) in the Administrative Controls Section of the Technical Specifications and the Relocation of the Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program,'' regarding the periodicity of dose projections for the calendar quarter and year.

Date of issuance: February 23, 2009.

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 156, 156, 161, 161, 184, 43, 230, 223, 190, 177, 197, 158, 272, 270, 274, 242, 237 and 268.

Facility Operating License Nos. NPF–72, NPF–77, NPF–37, NPF–66, NPF–62, DPR–2, DPR–19, DPR–25, NPF–11, NPF–18, NPF–39, NPF–85, DPR–16, DPR–44, DPR–56, DPR–29, DPR–30, and DPR–50: The amendments revised the Technical Specifications/Licenses.

Date of initial notice in ***Federal Register****:* May 20, 2008 (73 FR 29162). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 2009.

No significant hazards consideration comments received: No.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 8, 2008.

Description of amendment request: This amendment changes the Technical Specifications to delete Surveillance Requirement 4.6.3.1, which specifies post-maintenance testing requirements for containment isolation valves.

Date of issuance: February 23, 2009.

Effective date: As of its date of issuance, and shall be implemented within 90 days.

Amendment No.: 120.

Facility Operating License No. NPF–86: The amendment revised the License and Technical Specifications.

Date of initial notice in ***Federal Register****:* August 26, 2008 (73 FR 50361). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 2009.

No significant hazards consideration comments received: No comments were received. However, a hearing was requested which included contentions challenging the NRC staff's proposed no significant hazards consideration determination. On October 14, 2008, the request for hearing was denied by the Atomic Safety and Licensing Board. In accordance with 10 CFR 50.91(a)(3), the NRC staff made a final determination of no significant hazards consideration which is included in the Safety Evaluation.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit No. 1 (NMP1), Oswego County, New York

Date of application for amendment: February 25, 2008.

Brief description of amendments: The amendment revises NMP1 Technical Specification (TS) Section 3/4.4.4, ''Emergency Ventilation System,'' to remove the operability and surveillance requirements for the 10,000 watt heater located in the common supply inlet air duct for the Reactor Building Emergency Ventilation System. The amendment also revises TS 3/4.4.5, ''Control Room Air Treatment System,'' to reduce the 10-hour duration monthly system operational surveillance test requirement to a 15-minute run surveillance test requirement.

Date of issuance: February 17, 2009.

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 201.

Renewed Facility Operating License No. DPR–063: The amendment revises the License and TSs.

Date of initial notice in ***Federal Register****:* April 8, 2008 (73 FR 19110). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 17, 2009.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit No. 2 (NMP2), Oswego County, New York

Date of application for amendment: August 14, 2008.

Brief description of amendment: The amendment revises the NMP1 Technical Specification (TS) Surveillance Requirement frequency in TS 3.1.3, ''Control Rod Operability,'' and Example 1.4–3 in TS Section 1.4, ''Frequency,'' to clarify the applicability of the 1.25 test interval extension. The proposed changes are consistent with the Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF–475, ''Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,'' and NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 3.0. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on November 13, 2007 (72 FR 63935).

Date of issuance: February 23, 2009.

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 130.

Renewed Facility Operating License No. NPF–69: Amendment revises the License and Technical Specifications.

Date of initial notice in ***Federal Register****:* October 21, 2008 (73 FR 62567). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 2009.

No significant hazards consideration comments received: No.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: February 8, 2008, as supplemented by letter dated April 25, 2008, and email dated January 7, 2009.

Brief description of amendment: The amendment revises Technical Specification 5.6.6, ''Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),'' to include a new methodology for establishing reactor pressure vessel pressure-temperature limits in the Ginna PTLR. The new PTLR methodology is documented in WCAP–14040–A, Revision 4, ''Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.''

Date of issuance: February 23, 2009.

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No.: 106.

Renewed Facility Operating License No. DPR–18: Amendment revised the License and Technical Specifications.

Date of initial notice in ***Federal Register****:* April 8, 2008 (73 FR 19111). The supplemental letter dated April 25, 2008, and email dated January 7, 2009, provided additional information that clarified the application, did not expand the scope of the Application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated February 23, 2009.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: October 8, 2008.

Brief description of amendment request: The amendments revise the TS for the diesel fuel oil testing program. The proposed changes are based on NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF–374, revision 0. Prior notice of such a proposed change using the Consolidated Line Item Improvement Process was provided in the **Federal Register** on April 21, 2006 (71 FR 20735).

Date of issuance: February 20, 2009.

Effective date: As of the date of issuance and shall be implemented within 90 days from the date of issuance.

Amendment Nos.: 181 and 174.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revised the licenses and the technical specifications.

*Date of initial notice in **Federal Register**:* December 16, 2008 (73 FR 76413) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 20, 2009.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: December 1, 2008.

Brief description of amendment: On October 31, 2008, the NRC approved Amendment No. 186 to allow a one-time extension to the Completion Times for both essential service water (ESW) trains and the emergency diesel generators from 72 hours to 14 days. Amendment No. 186 was effective on the date of issuance and approved implementation by December 31, 2008, to permit replacement of ESW piping. The licensee completed the replacement of ESW Train A piping, but deferred the replacement of ESW Train B piping to early 2009. Amendment No. 191 authorized implementation of the ESW Train B piping prior to April 30, 2009.

Date of issuance: February 24, 2009.

Effective date: As of its date of issuance, and shall be implemented prior to April 30, 2009.

Amendment No.: 191.

Facility Operating License No. NPF–30: The amendment revised the Operating License and Technical Specifications.

*Date of initial notice in **Federal Register**:* December 23, 2008 (73 FR 78858).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 2009.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 26th day of February 2009.

For the Nuclear Regulatory Commission.

Joseph G. Giitter,

Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. E9–4898 Filed 3–9–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of March 9, 16, 23, 30, April 6, 13, 2009.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

Week of March 9, 2009—Tentative

There are no meetings scheduled for the week of March 9, 2009.

Week of March 16, 2009—Tentative

Monday, March 16, 2009

9:30 a.m.

Briefing on State of Nuclear Materials and Waste Programs (Public Meeting) (*Contact:* Tammy Bloomer, 301–415–1725).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Tuesday, March 17, 2009

1:30 p.m.

Briefing on State of Nuclear Reactor Safety Programs (Public Meeting) (*Contact:* Tammy Bloomer, 301–415–1725).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Friday, March 20, 2009

9:30 a.m.

Briefing on the Nuclear Education Program (Public Meeting) (*Contact:* John Gutteridge, 301–492–2313).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of March 23, 2009—Tentative

There are no meetings scheduled for the week of March 23, 2009.

Week of March 30, 2009—Tentative

There are no meetings scheduled for the week of March 30, 2009.

Week of April 6, 2009—Tentative

There are no meetings scheduled for the week of April 6, 2009.

Week of April 13, 2009—Tentative

Wednesday, April 15, 2009

9:30 a.m.

Briefing on NRC Corporate Support (Public Meeting) (*Contact:* Karen Olive, 301–415–2276).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Thursday, April 16, 2009

1:30 p.m.

Briefing on Human Capital and EEO (Public Meeting) (*Contact:* Kristin Davis, 301–492–2266).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings, call (recording)—(301) 415–1292. Contact person for more information: Rochelle Bavol, (301) 415–1651.

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/about-nrc/policy-making/schedule.html.*

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, Rohn Brown, at 301–492–2279, TDD: 301–415–2100, or by e-mail at *rohn.brown@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *darlene.wright@nrc.gov.*